HOLTEC BRITAIN

A Holtec International Company

Holtec Britain Ltd	HI-2240349
Sponsoring Company	**Document Reference**
0	30 September 2024
Revision No.	**Issue Date**
Report	Non-proprietary
Record Type	**Proprietary Classification**
ISO 9001	No
Quality Class	**Export Control Applicability**

Record Title:

PSR Part B Chapter 18 Structural Integrity

Proprietary Classification

This record does not contain commercial or business sensitive information.

Export Control Status

Export Control restrictions do not apply to this record.

Revision Log

Revision	Description of Changes
0	First Issue to Regulators

Table of Contents

List of Figures

No table of figures entries found.

List of Tables

18.1 INTRODUCTION

The Fundamental Purpose of the Generic Design Assessment (GDA) Safety, Security and Environment Case (SSEC) is to demonstrate that the Generic Small Modular Reactor (SMR)-300 can be constructed, operated, and decommissioned on a generic site in the United Kingdom (UK) to fulfil the future licensee's legal duties to be safe, secure and protect people and the environment as defined in Preliminary Safety Report (PSR) Part A Chapter 1 Introduction [1].

The Fundamental Purpose is achieved through the Fundamental Objective of the PSR, which is to summarise the safety standards and criteria, safety management and organisation, claims, arguments and intended evidence to demonstrate that the generic SMR-300 design risks to people are likely to be tolerable and As Low as Reasonably Practicable (ALARP) as defined in the GDA Scope Report [2].

Part B Chapter 18 of the PSR presents the Claims, Arguments and intended Evidence (CAE) for the approach to Structural Integrity for higher Reliability Structures, Systems and Components (SSCs) that underpin the design of the generic SMR-300.

18.1.1 Purpose and Scope

The Overarching SSEC Claims are presented in PSR Part A Chapter 3 CAE [3] of this PSR.

This chapter (Part B Chapter 18) links to the overarching claim through Claim 2.2:

Claim 2.2: The design of the systems and associated processes have been developed taking cognisance of relevant good practice and substantiated to achieve their safety and non-safety functional requirements.

As set out in Part A Chapter 3 [3], Claim 2.2 is further decomposed across several engineering disciplines which are responsible for development of the design of relevant SSCs. This chapter presents the Structural Integrity aspects for the generic SMR-300 and therefore directly supports a claim focused on the design and assessment of Structural Integrity SSCs, Claim 2.2.9.

Claim 2.2.9: Higher Reliability SSCs are justified using appropriate methods such that risk is tolerable and As Low as Reasonably Practicable (ALARP).

Further discussion on how the Level 3 claim is broken down into Level 4 claims and how the Level 4 claims are met is provided in Subchapter 18.3.

The Structural Integrity SSCs included within the scope of this chapter, are defined in GDA Scope Report [2], and can be found in Table 1.

This chapter covers codes and standards associated with the design of these SSCs and how they meet Relevant Good Practice (RGP) (subchapter 18.4), an overview of the methodology for identifying higher reliability components (subchapter 18.5), an introduction to the methodology for higher reliability demonstration (subchapter 18.6), Achievement of integrity (subchapter 18.7), Demonstration of integrity (subchapter 18.8), Monitoring (subchapter 18.9). Finally, a summary of considerations against the ALARP principle is provided, together with any Forward Actions (FAs) or commitments that have arisen (subchapter 18.10).

A master list of definitions and abbreviations relevant to all PSR Chapters can be found in PSR Part A Chapter 2 General Design Aspects and Site Characteristics [4].

18.1.2 Assumptions

No assumptions have been identified for this chapter.

18.1.3 Interfaces

The Structural Integrity discipline interfaces with multiple topic areas across the plant.

A description of the Structural Integrity SSCs of the generic SMR-300 reactor design is presented in PSR Part B Chapter 1 Description of the Reactor Coolant System and Engineered Safety Features [5], PSR Part B Chapter 2 Reactor Fuel and Core [6] and PSR Part B Chapter 5 Description of the Reactor Supporting Facilities [7].

Part A Chapter 3 [3] presents the high-level claims for the generic SMR-300 and provides a breakdown of claims for all chapters, including those within subchapter 18.3 of this chapter.

PSR Part A Chapter 5 Summary of ALARP' [8] provides the ALARP overview for the PSR and takes the ALARP summary from subchapter 18.8 of this chapter.

PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance (MSQA) [9] will assure that safety and quality tasks such as Examination, Inspection, Maintenance and Testing (EIMT) and Verification and Validation (V&V) are undertaken to the appropriate standard.

PSR Part B Chapter 9 Conduct of Operations [10]: The conduct of operations chapter details EIMT, ageing and degradation, which have expectations related to SI.

PSR Part B Chapter 10 Radiological Protection [11]: The SI of the reactor is vital for radiological protection. The ALARP justification for SI of the design in areas such as materials impacts Radiological Protection.

PSR Part B Chapter 12 Control of Non-Radiological Hazards [4]: Structural Integrity plays a role in mitigating mechanical failures and chemical leaks. Structural Integrity expectations for operations such as inspection impacts Nuclear Site Health and Safety captured in this chapter.

PSR Part B Chapter 13 Radioactive Waste Management [12]: The ALARP justification for the materials selected for use in the generic SMR-300 design will consider both Structural Integrity and Radioactive Waste Management (RWM).

PSR Part B Chapter 14 Design Basis Accident Analysis [13] : Structural Integrity of the reactor determines how the reactor would respond to potential accidents. Meanwhile, Design Basis Accident Analysis (DBAA) also provides input to the Structural Integrity chapter by identifying the conditions that the SSCs would be subjected to during a Design Basis Accident (DBA).

PSR Part B Chapter 15 Beyond DBA (BDBA), Severe Accidents Analysis and Emergency preparedness [14]: The Structural Integrity chapter interfaces with BDBA, Severe Accidents Analysis to assess extreme scenarios and their effects on Structural Integrity and enhance structural robustness against severe accidents.

PSR Part B Chapter 16 Probabilistic Safety Analysis [15]: Structural Integrity chapter provides input to the assessment of the containment structural analysis, as well as the Probabilistic Safety Analysis (PSA). It also affects the likelihood of various failure modes.

PSR Part B Chapter 17 Human Factors [16]: Structural Integrity interfaces with Human Factors to consider human interactions with SSCs during EIMT.

PSR Part B Chapter 19 Mechanical Engineering [17]: Systems such as the Reactor Coolant Circuit have both mechanical and metallic Structural Integrity aspects. This chapter supports the SI claims made for the mechanical SSCs.

PSR Part B Chapter 21 External Hazards [18]: Effects of external hazards are considered to ensure that Structural Integrity of SSCs is maintained throughout.

PSR Part B Chapter 22 Internal Hazards [19]: By considering the effects of internal hazards, not only is the Structural Integrity of SSCs maintained throughout their lifecycle, but it also contributes to the mitigation of these hazards.

PSR Part B Chapter 23 Reactor Chemistry [20]: The ageing and degradation owned by the Structural Integrity chapter will be cross cutting with Reactor Chemistry.

PSR Part B Chapter 25 Construction & Commissioning [21]: Structural Integrity is considered in the construction and commissioning programme to ensure that the SSCs are built to withstand the loads they will encounter during operation through suitable material selection, manufacturing methods and tests.

PSR Part B Chapter 26 Decommissioning Approach [22]: Structural Integrity of the SSCs influences the strategies and methods used in decommissioning.

18.2 SSCS WITHIN THE SCOPE OF STRUCTURAL INTEGRITY

Structural Integrity is a key element in demonstrating the safety of a new nuclear facility, it underpins and interfaces with several topics to provide a full safety case. The Office for Nuclear Regulation (ONR) Safety Assessment Principles (SAPs) [23] place particular scrutiny on the demonstration of safety for the Structural Integrity of metallic components and introduces a classification above that for other topic areas i.e., higher reliability.

While the classification of SSCs is explained in PSR Part A Chapter 2 [4], the need for identification of SSCs with higher reliability claims is driven by the requirements of Structural Integrity. Therefore, an overview of the requirements is provided along with a summary of SSCs within the scope of Structural Integrity.

18.2.1 Structural Integrity Classification

The safety categorisation and classification methodology currently defined for the SMR-300 by Holtec International utilises the United States Nuclear Regulatory Commission (US NRC) Regulatory Guide 1.26, Revision 6 [24] and related guidance documents.

The design of the generic SMR-300 will require the adoption of appropriate national and international nuclear specific codes and standards to meet regulatory expectations so that it is fit for use as the starting point for a future licensee's site-specific project. Holtec acknowledge the existence of differences in the approach to safety categorisation and classification between the NRC Regulatory Guides [25] and other national and international standards.

The differences in the approach to safety categorisation and classification have been identified via the UK GDA Gap Analysis Report [26] and the US/UK Regulatory Framework and Principles Report [27]. Work will be conducted moving forward to address all gaps identified in the SSEC Revision 0 relating to categorisation and classification to ensure regulatory expectations are met.

In line with the SAPs [23], it is recognised that there may be SSCs within the generic SMR-300 that have highest reliability claims. These higher reliability SSCs are a key part of the safety categorisation and classification methodology. Methodology for Determining Higher Reliability Classification [28] details the proposed methodology for identification of higher reliability SSCs.

18.2.1.1 Higher Reliability SSCs

Higher reliability SSCs are identified as two separate Structural Integrity classes (see subsection 18.5):

- Very High Reliability (VHR).
- High Reliability (HR).

The SSCs in the table below have been highlighted as candidate higher reliability components:

Table 1: (REDACTED)

REDACTED

Consequence analysis will inform the further classification of these candidate VHR/HR SSCs. The assignment of higher reliability claims will be refined based on the role of a system, component, or region in maintaining nuclear safety against significant radiological consequences, such as weld regions.

18.2.1.2 Higher reliability design philosophy

Where possible, the application of a higher reliability claims should be avoided. It is preferential to the generic SMR-300 that an engineered defence-in-depth solution is provided, but where this is deemed not practicable, then a higher reliability claim will be explored. It is understood that a high burden of proof is attached to the higher reliability claim and for the safety case to be acceptable, beyond codes measures must be undertaken.

18.3 STRUCTUAL INTEGRITY CLAIMS, ARGUMENTS AND EVIDENCE

The primary purpose of a CAE approach is to capture the golden thread of a safety case narrative demonstrating how plant and operational evidence is brought together to justify that a high-level or fundamental claim is true. In the context of the generic SMR-300, that is how the Fundamental Purpose of the SSEC (presented in Part A Chapter 2 [4]) is achieved.

The Fundamental Purpose follows a golden thread throughout the SSEC to CAE via the objectives of the PSR, Preliminary Environmental Report (PER), and Generic Security Report (GSR) [29]. The overarching SSEC claims are presented in Part A Chapter 3 [3], and this chapter links to the overarching claims through Claim 2.2:

Claim 2.2: The design of the systems and associated processes have been developed taking cognisance of relevant good practice and substantiated to achieve their safety and non-safety functional requirements.

This chapter presents the Structural Integrity topic for the generic SMR-300 to support Claim 2.2.9:

Claim 2.2.9: Higher Reliability SSCs are justified using appropriate methods such that risk is tolerable and As Low as Reasonably Practicable (ALARP).

Claim 2.2.9 has been further decomposed within Part B Chapter 18, to provide confidence that the relevant requirements on structures will be met during all lifecycle phases.

This has been done by breaking down Claim 2.2.9 into four further sub-claims.

Claim 2.2.9.1 contributes to the design phase by defining the codes and standards that the design will be assessed against.

Claim 2.2.9.2 ensures that Structural Integrity SSCs achieve integrity using best practice engineering methodologies.

Claim 2.2.9.3 ensures Structural Integrity SSCs demonstrate integrity through defect tolerance assessment and qualified inspection.

Claim 2.2.9.4 covers through-life monitoring of the plant during operation. This assures that the plan operates within the prescribed boundaries and that degradation is monitored. Any deviation for expected degradation and aging will be captured using diverse and robust systems.

Table 2 shows in which chapter of this PSR these claims are demonstrated to be met.

Table 2: CAE Chapters

Claim No.	Claim	Chapter Section
2.2.9.1	Structural Integrity SSCs are designed using appropriate Codes and Standards.	18.4 Codes and Standards
2.2.9.2	Higher reliability SSCs Structural Integrity is achieved through quality design, analysis, manufacture, and testing.	18.7 Achievement of integrity
2.2.9.3	Higher reliability components will be tolerant of defects demonstrated by the avoidance of fracture.	18.8 Demonstration of integrity
2.2.9.4	Through life monitoring, maintenance and inspection will provide forewarning of failures.	18.9 Monitoring

A summary of the current CAE route map for Part B Chapter 18 is provided in Appendix A Table 4 which is taken from the generic SMR-300 Overarching SSEC Claim Route Map presented in Appendix A of Part A Chapter 3 ‘CAE’ [3]. A further update on claim decomposition, argument development and evidence maturity will be provided in the subsequent update of the chapter.

18.4 CODES AND STANDARDS

Claim 2.2.9.1: Structural Integrity SSCs are designed using appropriate Codes and Standards.

This subchapter outlines the codes and standards used in the design of the generic SMR-300 Structural Integrity SSCs.

The Requesting Party (RP) has recognised that UK nuclear safety regulations are based on a non-prescriptive regime and consequently the technical codes and standards that must be used for nuclear power plant are not prescribed. However, codes and standards used should reflect the industry practice and comply with UK regulation. New codes and standards can be introduced where needed. Use of such codes will be justified in each case.

18.4.1 Codes and Standards

The design philosophy for safety-related items within the scope of Structural Integrity is to design in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III [30]. Non-safety-related items are to comply with the design requirements stipulated by ASME BPVC Section VIII, along with any other relevant industry or manufacturer codes and standards. The principal regulations, codes, and standards are presented in Table 3.

Table 3: Principal Reference Regulations, Guides, Series, Codes, and Standards

Label	Title	Important Sections & Parts	Revisions
ASME BPVC	Boiler and Pressure Vessel Code [30]	• Section II, "Material" • Section III, "Rules For Construction of Nuclear Facility Components" • Section V, "Non-destructive Examination" • Section VIII, "Rules for Construction of Pressure Vessels" • Section IX, "Qualification Standard for Welding" • Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components"	2023 (2021 for Section III Engineering)
ASME B16	Standardization of Valves, Flanges, Fittings, and Gaskets [31]	• B16.5, "Pipe Flanges and Flanged Fittings" • B16.34, "Valves-Flanges, Threaded, and Welding End"	2017 2020
ASME B31	Pressure Piping [32]	• B31.1, "Power Piping" • B31.3, "Process Piping"	2022
ASME NQA-1	Quality Assurance Requirements for Nuclear Facility Applications [33]	Not applicable	2015
ASTM E185-82	Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels [34]	Not Applicable	July 1982
U.S.NRC 10 CFR	Title 10, Code of Federal Regulations [35]	• Part 21, "Reporting of Defects and Noncompliance," • Part 50, "Domestic Licensing of Production and Utilization Facilities" with pertinent appendices	Not Applicable
U.S.NRC NUREG-0800	Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition [36]	SRP 3.2.2, "System Quality Group Classification", and more relevant SRPs	2016

Label	Title	Important Sections & Parts	Revisions
U.S.NRC RG	Regulatory Guides [25]	1.20, 1.26, 1.28, 1.43, 1.45, 1.84, and more relevant guides	Not Applicable
TEMA	Standards of the Tubular Exchanger Manufacturers Association [37]	Not Applicable	2019 (10th Edition)

18.4.2 CAE Summary

The generic SMR-300 design has been developed using nuclear specific codes and standards. These codes and standards include guidance and expectations from the US NRC and represent application of relevant good practice. Claim 2.2.9.1 has therefore been met to the extent consistent with the maturity of this project at this time; further work will be undertaken and reported in the next revision of this PSR

18.5 METHODOLOGY FOR IDENTIFYING HIGHER RELIABILITY COMPONENTS.

A review of the ONR expectations on Structural Integrity, lessons learnt from the previous GDA, and the UK RGP has informed the development of identifying and assigning highest reliability claims on SSCs, which is presented in the Methodology for Determining Higher Reliability Classification [28].

At this stage, higher reliability candidates have been identified with respect to their preliminary consequences of gross failure. As the GDA progresses, the postulated gross consequences of failure of SSCs will be further investigated to inform the identification of candidate higher reliability SSCs. These candidate SSCs, highlighted in Table 1.

, would have a primary safety function for maintaining plant safety and their direct or in-direct consequences of gross failure may result in unacceptable/highly undesirable conditions with release of radioactivity, due to lack of adequate physical protection.

The candidate higher reliability SSCs will be evaluated with a more detailed consequence analysis to finalise the classification. The SSCs will be assigned the higher reliability claims when the consequences of gross failure can't be demonstrated to be acceptable, and it is not reasonably practicable to provide further defence-in-depth with lines of physical protection. The severity of consequences and available physical protection will distinguish the classes of the components.

The higher reliability SSCs are identified as two separate Structural Integrity classes above Class 1:

- Very High Reliability.
- High Reliability.

18.5.1.1 Very High Reliability Components

VHR is assigned to components for which the consequences of gross failure would be catastrophic and unacceptable. These components have no line of protection against the release of radioactive material, and it is not reasonably practicable to provide physical defence-in-depth. High levels of evidence are required to substantiate the higher reliability claims for these components based on sound engineering and beyond design code assessments, thus enabling a claim that the corresponding failure frequency is less than 10^{-7} per annum in line with [38].

18.5.1.2 High Reliability Components

HR claims are assigned when the gross failure of a component would result in "highly undesirable" consequences. For these components, there is limited protection to prevent unacceptable levels of radiological release but still would result in "highly undesirable" consequences**.** Appropriate level of demonstration will be provided in support of the safety case by introducing additional measures beyond normal practice compared to SC1 components. However, compared to VHR, a lower range of evidence provision would be adequate to substantiate the claims for HR, as the consequences of gross failure are less severe. Similarly, the probability of failure of HR components would be expected to be in the region between 10^{-5} and 10^{-7} per annum.

18.6 HIGHER RELIABILITY DEMONSTRATION

The established nuclear codes referred to in subchapter 18.4.1 describe the normal practices for components with SI claims and are accepted as the minimum specifications to the SSCs with highest reliability claims. However, it is acknowledged that these recognised design code assessments can't be used to discount gross failure of an SSC.

Therefore, a safety case to discount gross failure calls for additional measures beyond normal practices that would highlight conceptual defence-in-depth where it is not reasonably practicable to provide a physical defence in depth. The beyond code demonstration is not required by the US regulatory body. This work will be undertaken moving forward to meet the UK regulatory expectations.

Avoidance of fracture will be demonstrated through conservative Defect Tolerance Assessment (DTA), qualified manufacturing inspections, and confirmatory fracture toughness testing. This demonstration will be applied to the selected limiting locations of VHR/HR components.

To establish highest reliability safety case, conceptual defence-in-depth will be presented by the multi-legged approach advised by UK Technical Assessment Group Structural Integrity of High Integrity Plant (TAGSI) [38]. This will allow provision of a range of arguments that will substantiate the highest reliability claims and enable a failure rate claim of less than 10^{-7} per annum. The three legs of UK TAGSI arguments are,

- Leg A. Sound design and manufacture based on Operating Experience (OPEX) and Functional testing.
- Leg B. Failure Analysis.
- Leg C. Forewarning of failure.

18.7 ACHIEVEMENT OF INTEGRITY

Claim 2.2.9.2: Higher reliability SSCs Structural Integrity is achieved through quality design, analysis, manufacture, and testing.

This subchapter outlines the engineering analysis methodologies used in the design of the generic SMR-300 Structural Integrity SSCs. It considers:

- Design;
- Operational Limits and Design Parameters;
- Materials;
- Analysis;
- Quality Assurance;
- Manufacturing and Installation;
- Welds;
- Examination, Inspection, Maintenance and Testing;

18.7.1 Design

Holtec International have an extensive inhouse capability to manufacture, load test, conduct Non-Destructive Examination (NDE), performing analyses and many other areas key to designing a Nuclear Power Plant (NPP). Having in-house capability and established processes results in a well-rounded approach to design for X through experience in, manufacturing, design, analysis, inspection, and validation. This process, as well as existing procedures, drives informed design decisions when considering inspection accessibility, component material selection or weld minimisation and arrangement.

During the design of the generic SMR-300, Holtec International's organisational experience has been applied to the SSC design process considering their future pre-service and In-Service Inspections (ISI), including access arrangements and geometry allowing for effective NDE. As the generic SMR-300 progresses through the defined and stage gated design process, regard will be given to the positioning, size, number, and environment of welds. Considering ISI, the SSCs are designed for inspectability of welds and where possible the minimisation of welds is balanced appropriately with the challenges of manufacturing large forgings. This minimisation of vessel welds and reduced number of components results in a simplified design.

OPEX of known technical issues, degradation and established design codes are all utilised in order to inform design decisions. The SSCs have been designed to produce failure modes that are gradual and predictable, while avoiding brittle behaviour, as per Generic Design Criteria (GDC) 31 [35]. For example, the RPV beltline section doesn't have welds in the high neutron fluence regions to reduce possibility of irradiation embrittlement. The pressure retaining components and any other support structures, in the scope of SI, are designed in line with the corresponding ASME BPVC, Section III class requirements [30] following the guidance by US NRC NUREG-0800 Standard Review Plan 3.2.2 [39].

The design of components has been investigated for the different service levels with a suitably conservative combination of loads. Loads are combined in line with the NUREG-0800 Standard Review Plan (SRP) 3.9.3 [36] and ASME BPVC to calculate the stress limits. This combination may include effects of internal pressure, dead weight, thermal expansion, dynamic loads due to seismic motion. Conservative assumptions are made for both load values and frequency of events combined.

While the design of SSCs has been conducted to the US regulatory guidance, there is no formal process that documents why particular design decisions are ALARP. As evidencing ALARP is a UK regulatory requirement this will be addressed as the GDA progresses.

18.7.2 Operational limits and design parameters

Components and structures will be designed and analysed within specific defined limits. Therefore, they should be operated and controlled within these limits and conditions. Throughout the plant's service life, operating conditions will be monitored to ensure they are maintained within these limits.

Postulated events that the plant might credibly experience during design life and subsequent maintenance activities are considered, to establish the design condition for SSCs with SI claims. The design transients, along with the consequent loads and load combinations, will be evaluated in accordance with the appropriate ASME code service limits. This ensures that the SSCs are designed to remain within acceptable limits to accommodate the pressures and temperatures anticipated during normal operation and accident loading conditions such as Safe Shutdown Earthquake (SSE).

SSCs have been designed with appropriate safety margins to ensure their structural integrity, taking into account service lifetime. The overpressure protection features have been designed with enough capacity to prevent the Reactor Coolant Pressure Boundary (RCPB) from exceeding 110% of design pressure during Normal Operations (NOs) and Anticipated Operational Occurrences (AOOs).

18.7.3 Materials

The material selection for the generic SMR-300 will be carried out through a systematic approach by considering the behaviour of equipment in the manufacturing, operation, inspection, and maintenance stages, as well as previous OPEX in similar environments. The evaluation of material suitability will involve characterising the applicable environment, identifying potential degradation modes, and assessing potential hazards that could impact the continued effectiveness of the selected material. For instance, an optioneering process has been followed for the materials of reactor internal components, considering water chemistry, irradiation embrittlement and operational modes with temperature and pressure transients.

The use of materials for the generic SMR-300 components is informed by precedence set by existing designs, such as ESBWR and AP1000. Most of the reactor internals material shall be austenitic steel, whose corrosion-resistant characteristics are well established in the nuclear industry. Any materials other than austenitic stainless steel used within the reactor internals and core supports will be chosen to address specific needs such as strength, aversion to fatigue, and/or low corrosion susceptibility. The reactor internals and core support structures will be designed only with materials allowed in ASME BPVC, Section III, Subsection NG-2120 [40], and weld filler materials listed in ASME BPVC, Section II [41] will be used. The selection of the materials used for the reactor internals and placement of welds will consider the estimated peak neutron fluence to which the materials will be subjected. The reactor internals' materials will be evaluated for susceptibility to known ageing and degradation mechanisms.

These susceptibilities include irradiation-assisted stress corrosion cracking and void swelling, which will be addressed in the reactor internals material reliability programmes. The unstabilised austenitic stainless-steel components will be processed, inspected, and tested to minimise susceptibility to intergranular corrosion caused by sensitisation in line with US NRC RG 1.44 [42]. Compliance with US NRC RG 1.28 [43] and RG 1.33 [44] will limit introduction of contaminants for tools used in abrasive work on austenitic stainless-steel surfaces.

When selecting the material for piping, corrosion allowance is allocated based on industry experience for the life cycle of the NPP. The following areas are then considered to arrive at an informed decision, working fluid quality, material composition, and welding preheat and Post Weld Heat Treatment (PWHT) [45]. A Long-term maintenance strategy should also be in place for the piping consisting of inspection and repair or replacement of parts.

18.7.4 Analysis

The central objective of the Structural Integrity Analyses is to ensure that the generic SMR-300 plant possesses sufficient structural capability to withstand normal, off-normal loads, and the worst-case loadings under extreme environmental phenomena or accident events. The input parameters to these analyses are suitably conservative, to provide an adequate margin for uncertainty. These inputs include conservative material properties that will be confirmed by representative testing.

SSC classification will be used to identify the appropriate level of analysis. The classification of the SSC will determine analysis to the accepted codes and standards. Where higher reliability claims are made, beyond code methods of analysis will be undertaken.

The Reactor Coolant System will be designed in line with 10 CFR 50 Appendix G [46] and therefore ASME BPVC Section III, Nonmandatory Appendix G requirements [40]. ASME BPVC Section III Sub-section NB and Appendix XIII [40] will be used to investigate plastic collapse, local failure, buckling, and cyclic loading of the system.

Holtec has a comprehensive internal procedure for V&V of computer codes and calculation methods. V&V problems and sensitivity studies will be applied to computer codes, such as R-CODE module for R6 [47], that are used to investigate integrity of SSCs of the generic SMR-300.

18.7.5 Quality Assurance

With reference to Part A Chapter 4 (MSQA) [9], the Quality Assurance (QA) requirements will be applied to the design, procurement, manufacture, operation, and testing to ensure the safety-related work is performed in accordance with approved QA procedures as described in the Topical Report on The Quality Assurance Program (QAP) for Holtec International’s SMR Design and Construction [48]. This will ensure that inputs to Structural Integrity are accurate.

During the design process, measures are in place to manage inputs, outputs, changes, interfaces, and records within Holtec and its suppliers, subject to the provisions of the QAP [48] . These controls ensure that design inputs are accurately translated into design outputs, with the final design output referencing suitable acceptance criteria that allows for verification through inspection and testing as necessary.

Necessary measures and governing procedures have been established to control procurement, ensuring compliance with defined requirements. The interface requirements for procured equipment and supports will be provided to the supplier in the purchase order. Holtec commits to requirements of ASME NQA-1 2015 for the control of purchased material, equipment, and services with an inspection program to verify their quality, as outlined in QAP [48]. All safety-related materials require a certified material test report, alongside an independent review for conformance to the specification.

Holtec will establish and implement measures to assess the quality of purchased items and services, whether purchased directly or through contractors, at intervals and to a depth consistent with the item or service importance to safety, complexity, quantity, and the frequency of procurement. Verification actions may include testing, as appropriate, during design activities. Verifications occur at the appropriate phases of the procurement process, and may include, as necessary, verification of activities of suppliers.

18.7.6 Manufacturing and Installation

The manufacturing and installation process for ASME BPVC, Section III components will be largely guided by Section III and IX , as they provide the baseline requirements following the US NRC NUREG-0800 SRP 3.2.2 [39] guidance on codes and standards application on classes. Where no advantage can be taken from novel technological advancements, only proven techniques will be employed, using approved and controlled procedures in line with ASME NQA-1 requirements. Where HR and VHR components are identified, appropriate beyond code measures will be undertaken in compliance with the guidance provided by the UK regulatory body.

Holtec International's background in manufacturing and installation gives them a wealth of experience with utilising RGP and OPEX. The manufacturing methodology adopted will facilitate examination during the manufacturing of SSCs. At agreed hold points, independent inspections will be conducted, of HR and VHR components.

18.7.6.1 Welds

Welds in the generic SMR-300 plant will be categorised based on their safety classification to ensure that all welds shall perform their intended function, the following requirements will be conformed to:

- The weld joint configuration is selected in accordance with the function of the joint.
- The welding procedure specifications comply with ASME BPVC, Section IX [49] for every Code material used in the system.
- The quality assurance requirements applied to the welding process correspond to the highest safety classification of the parts being joined.
- The non-destructive examination of every code weld is carried out using quality procedures that comply with ASME BPVC, Section V [50].

The welding operations are performed in accordance with the requirements of codes and standards depending on the design and functional requirements of the components. The welder of the non-structural welds will be qualified with ASME BPVC, Section IX, or AWS D1.1 [51]. The controls on all ASME BPVC welds will be based on Section III.

To supplement the above process, where welds are classified as HR or VHR, beyond code checks are required and will be follow the structure in subchapter 18.8.1.2.

18.7.7 Examination, Inspection, Maintenance and Testing

An examination strategy is developed to assure adequate manufacturing of components has been achieved. The defect size that can be confidently measured by the selected examination method will be used as an input to the DTA along with a factor, assuring margins between the acceptable defect size and the ability to detect the defect. The NDE of the base materials used for the reactor internals and core support materials will be in accordance with the ASME BPVC, Section III, Division I, NG-2500 [40]. Similarly, the acceptance criteria for the NDE methods of weld materials will be in accordance with the requirements of the ASME BPVC, Section III, Division 1, NG-5000 [40]. Where beyond code measures are required, an enhanced examination standard will be utilised to provide the extra assurance. This qualified inspection is discussed in more detail in 18.8.1.2

Holtec have formalised programs with criteria determining when testing is required to demonstrate that performance of equipment and plant systems is in accordance with the design parameters. These tests will be in accordance with applicable procedures. Test results will be documented and evaluated to assure that the test requirements have been satisfied. Test programs will ensure appropriate retention of test data in accordance with the records requirements of Holtec's QAP. Further description of the measures applied for inspection are provided in the CD-03, Nuclear Quality Assurance Manual [52] and subordinate procedures Topical Report [48].

Where Leak Before Break (LBB) has been claimed, EIMT will be undertaken to verify as-built material specifications for piping including, toughness (J-R curves), tensile strength (stress-strain curves), yield and ultimate strength, and welding process/methods used to ensure that the actual plant-specific stress analysis based on the above inputs satisfy the bounding LBB analysis. The US deployment of LBB differs from the way that this can be applied in the UK. In the UK where higher reliability is claimed, LBB is not appropriate as the main safety case argument. However, LBB is recognised as a supporting argument which can be used to aid the demonstration of conceptual defence in depth.

18.7.8 CAE Summary

The achievement of integrity claim has drawn upon the comprehensive scope and application of the ASME BPVC to demonstrate how it will be evidenced. The claim recognises key differences between the approach prescribed by the NRC and those areas to be addressed in the UK context. Further work to address these differences will continue as the GDA progresses. Claim 2.2.9.2 has therefore been met to the extent consistent with the maturity of this project at this time; further work will be undertaken and reported in the next revision of this PSR.

18.8 DEMONSTRATION OF INTEGRITY

Claim 2.2.9.3: Higher reliability components will be tolerant of defects demonstrated by the avoidance of fracture.

This subchapter outlines the approach to DTA to be used on the generic SMR-300 Structural Integrity SSCs. It considers:

- Avoidance of Fracture Demonstration;
    - Defect Tolerance Assessment
    - Qualified Inspection
    - Testing for Fracture Toughness

18.8.1 Avoidance of Fracture Demonstration

18.8.1.1 Defect Tolerance Assessment

For the avoidance of fracture demonstration in VHR/HR SSCs, conservative DTA will be conducted through the R6 approach [47]. The R6 code for DTA has been through Holtec's QA Validation [53] and will ensure adequate margin is achieved between concerning defects and the defect size that can be detected confidently by the qualified inspections. The lower bound material properties will be calculated based on ASME BPVC Section III, Non-mandatory Appendix G requirements. The values used will be confirmed to be achievable and conservative through fracture toughness testing.

18.8.1.2 Qualified Inspection

Inspections are fundamental for establishing that the components are as defect free as possible, especially for welds which have a higher likelihood of defect occurrence. The manufacturing inspections of higher reliability SSCs will be qualified with an appropriate arrangement such as European Network for Inspection and Qualification (ENIQ) methodology [54]**.** This qualification will allow detection of defects of concern by NDE with higher confidence. The inspection scheme will include the qualification of higher reliability PSI/ISI and manufacturing NDE of selected higher reliability locations.

18.8.1.3 Testing for Fracture Toughness

As described in 18.8.1.1, conservative fracture toughness properties will be used for DTA of VHR/HR components. The values used will be confirmed to be lower bounds by representative fracture toughness tests. The methodology of testing weld and parent material properties of VHR/HR components will be reported.

18.8.2 CAE Summary

The demonstration of integrity claim has recognised the need for further work to evidence that it has been successfully achieved. The claim outlines the planned approach to DTA using appropriate methodologies. Claim 2.2.9.3 has therefore been met to the extent consistent with the maturity of this project at this time; further work will be undertaken and reported in the next revision of this PSR.

18.9 MONITORING

Claim 2.2.9.4: Through life monitoring, maintenance and inspection will provide forewarning of failures.

This subchapter outlines the approach to be used in the monitoring and in-service inspection of the generic SMR-300 Structural Integrity SSCs. It considers:

- Plant monitoring
- In-Service Inspection
- Environmental Material Surveillance

18.9.1 Plant Monitoring

Monitoring systems will be in place to provide the operating status of metallic components and structures in the generic SMR-300. Vibration, expansion, and contraction of the SSCs will be monitored to verify that it is within the bounding conditions identified in the analyses.

The primary monitoring systems that will provide assurance on maintaining Structural Integrity and forewarning of failure are:

- **The Leak Detection System (LDS),** which will be implemented to monitor the Reactor Coolant System for any leak and forewarn further safety significant failures and consequences.
- **Vibration Monitoring System (VMS),** which measure vibration in locations with high cycle fatigue risk.
- **Temperature Monitoring**, which will collect reactor coolant temperature data to inform Embrittlement Trend Curves.

18.9.2 In-Service Inspection

The reactor coolant pressure boundary components are designed with access for periodic inspection and testing of important areas and features to assess their structural and leak tight integrity. The requirements for the periodic inspection and testing of ASME BPVC, section III are in accordance with ASME BPVC, Section XI Division 1 [55] pursuant to 10CFR 50.55a(g). Where the initial manufacturing has been inspected and the process qualified to ENIQ or an equivalent, this initial fingerprint is compared to the results of the in-service qualified inspection and records kept for future inspections.

18.9.3 Environmental Material Surveillance

An optimal Material Surveillance Program, in line with 10CFR50 Appendix H [56], will allow monitoring of RPV material of the generic SMR-300, abiding by the rules of ASTM E185-82 [34] and thus changes in fracture toughness properties. This will include testing of the RPV beltline specimens (concerning “limiting” materials) located in the irradiation specimen baskets. This will provide representative knowledge with regards to the ageing and degradation of RPV material and inform/confirm the analysis.

18.9.4 CAE Summary

The maintenance of integrity claim recognises the systems and processes which will contribute towards the forewarning of component failures. The evidence to support this claim also draws upon the relevant good practice defined in the ASME BPVC. Claim 2.2.9.4 has therefore been met to the extent consistent with the maturity of this project at this time; further work will be undertaken and reported in the next revision of this PSR.

18.10 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP

This subchapter provides an overall summary and conclusion of the Structural Integrity Chapter and how this Chapter contributes to the overall demonstration of ALARP for the generic SMR-300. Part A Chapter 5 (PSR 'ALARP Summary' [8]) sets out the overall approach for demonstration of ALARP and how contributions from individual Chapters are consolidated.

This subchapter therefore consists of the following elements:

- Technical Summary;
- ALARP Summary;
    - Review against RGP;
    - Demonstration Against Risk Targets;
        - Evaluation of Risk;
    - Risk Reduction Options;
    - GDA Commitments and Forward Actions.
- Conclusion.

A review against these elements is presented below under the corresponding headings.

18.10.1 Technical Summary

PSR Part B Chapter 18, Revision 0 demonstrates that the Structural Integrity SSC within the scope of this report will meet the high-level Claims of the SSEC and that the SSC can be substantiated at Pre-Construction Safety Report (PCSR) stage. This is demonstrated through the following sub-claim:

Claim 2.2.9: Higher Reliability SSCs are justified using appropriate methods such that risk is tolerable and As Low as Reasonably Practicable (ALARP).

The design of Structural Integrity SSCs has been undertaken using best practice nuclear industry codes and standards by use of the ASME III BPVC design codes.

Material properties testing and transient definitions ensure that data used during code assessment will be appropriate and suitably conservative.

Avoidance of fracture will be demonstrated through conservative DTA, qualified inspections, and confirmatory fracture toughness testing.

The quality assurance requirements for Structural Integrity SSCs are defined in the Design Specifications for the respective SSCs.

An Material Surveillance Programme will allow monitoring of materials and thus changes in fracture toughness properties. This will include testing of the RPV beltline specimens located in the irradiation specimen baskets. This will provide representative knowledge with regards to the ageing and degradation of RPV material and inform the analysis.

PSI/ISI of Structural Integrity SSCs will be undertaken to ensure that any defects are within tolerance and that crack growth limits are not exceeded during operation.

PSR Revision 0 will define the hazards considered in the design and their magnitude by reference to PSR Part B Chapter 14, 'DBAA' [13], Part B Chapter 15, BDBA, Severe Accidents Analysis and Emergency Preparedness [14], PSR Part B Chapter 21, External Hazards [18] and PSR Part B Chapter 22, Internal Hazards [19]. It will present a summary of the Structural Integrity analysis undertaken for the generic SMR-300 and will identify any gaps.

18.10.2 ALARP Summary

18.10.2.1 Demonstration of RGP

The design of the generic SMR-300 Structural Integrity SSCs complies with RGP and US NRC requirements applicable in the US. The design adopts nuclear-specific codes and standards endorsed by the US NRC and internationally recognised bodies such as the International Atomic Energy Agency (IAEA). The principal codes and standards identified within subchapter 18.4 are considered RGP by the UK nuclear industry. This is based on existing practices adopted on UK nuclear licensed sites, application in earlier and successful GDAs, as well as recognition as RGP by ONR TAGs.

DTA is not undertaken in the US but is a UK regulatory requirement. DTA will be undertaken using the R6 approach [47]. The R-CODE R6 module for DTA has been through QA Validation [53] and will ensure adequate margin is achieved between concerning defects and the defect size that can be detected confidently by the qualified inspections. The use of the R6 approach is considered RGP and has been utilised for the safety cases of both operating UK reactors and as part of recent GDA's.

Forward actions will form the basis for setting out the process to justify any gaps from UK RGP. Forward Actions have been collated and are managed via the process described in PSR Part A Chapter 4 Lifecycle MSQA [9].

18.10.2.2 Demonstration Against Risk Targets

The numerical targets against which the demonstration of ALARP is considered can be found in PSR Part A Chapter 2 [4]. Structural Integrity SSCs, through the defined safety functions, will contribute to the demonstration of ALARP by comparison against the risk targets in two ways:

- By fulfilling safety functions for normal operations (e.g., shielding and containment), and thereby contributing to achieving Targets 1-3;
- By achieving their target probability of failure they will contribute to the achievement of accident risk, Targets 4-9.

Risks below the Basic Safety Objectives (BSOs) are considered broadly acceptable; however, the RP is still required to identify further risk reduction measures in line with the ALARP approach. Risks between the BSOs and Basic Safety Levels (BSLs) require a consideration of risk reduction options. Risks above the BSLs are not acceptable for new plants.

18.10.2.2.1 Evaluation of Risk

Evaluation of risk is not applicable to the Structural Integrity SSCs. The safety classification of the Structural Integrity SSCs will be associated with a probability of failure, which is then used to calculate the overall comparison against the risk targets as described above.

At this time, the evaluation of the normal operations and accident risks against Targets 1-9 has not been provided. This information will be presented in PSR Part B Chapter 10 'Radiological Protection' [11] for normal operations, and Part B Chapter 14 'Design Basis Accident Analysis' [13], Part B Chapter 15 'Beyond Design Basis and Severe Accident Analysis, and Emergency Preparedness' [14] Part B Chapter 16 'Probabilistic Safety Analysis' [15] for accident conditions.

18.10.2.3 Risk Reduction Options

This is a placeholder to identify and review relevant Position Papers and Design Decision Papers with a view to demonstrate which option(s) is/are ALARP.

It will summarise those option evaluations, and it will briefly explore if other risk reduction options have or could be considered and either:

- Present the ALARP argument for why those options have not been implemented.
- Present the ALARP argument for why those options will be implemented in future.
- Create a Forward Action to consider the option(s) at some future point (noting this still must be a point where a meaningful design improvement could be made).

The process for the assessment of risk reduction options is presented in 'HPP-3295-0017-R0, GDA Reference Design Process and GDA Prospective Design Change Register' [57]. Part A Chapter 5 of this PSR 'ALARP Summary' [8] considers the holistic risk-reduction process for the generic SMR-300.

18.10.2.4 GDA Commitments and Forward Actions

There are no GDA commitments identified for Part B Chapter 18, Structural Integrity.

Forward Actions have been collated and are managed via the process described in PSR Part A Chapter 4, 'Lifecycle MSQA' [9]. PSR Part A Chapter 5 'ALARP Summary' [8] describes the contribution of the forward actions to the ALARP argument.

18.10.3 Conclusion

The conclusion of this Chapter of the PSR is that:

- The Chapter Claims identified have been met to a maturity aligned with a preliminary safety report. Further claims, arguments and evidence will be presented in due course as the design develops.
- Safety functions have been identified for the candidate higher reliability SSCs.
- A systematic classification system will be applied to the Structural Integrity SSCs commensurate with their importance. The classification system allows the appropriate design codes and standards to be identified.

- The Structural Integrity SSCs have been designed using US codes and standards. Where required, beyond code measures will be undertaken utilising the TAGSI multi-legged approach to a safety case to demonstrate 'Conceptual' Defence in Depth.
- DTA will be undertaken utilising conservative material properties, fracture toughness testing, and qualified inspections to demonstrate the avoidance of fracture.
- Through-life inspections and monitoring are used to provide forewarning of failure ensuring that monitoring methods are appropriately diverse and robust.
- PSR Part A Chapter 5 ‘ALARP Summary’ [8] concludes that it can be demonstrated that the generic SMR-300 reduces risks to ALARP and that the Fundamental Purpose of the SSEC has been fulfilled.

18.11 REFERENCES

[1] Holtec Britain, "HI-2240332, Holtec SMR GDA PSR Part A Chapter 1 Introduction," Revision 0, August 2024.

[2] Holtec Britain, "HI-2240121, GDA Scope Report," Revision 0, March 2024.

[3] Holtec Britain, "HI-2240334, Holtec SMR GDA PSR Part A Chapter 3 Claims, Arguments and Evidence," Revision 0, August 2024.

[4] Holtec Britain, "HI-2240333, Holtec SMR GDA PSR Part A Chapter 2 General Design Aspects and Site Characteristics," Revision 0, August 2024.

[5] Holtec Britain, "HI-2240337, Holtec SMR GDA PSR Part B Chapter 1 Reactor Coolant System and Engineered Safety Features" Revision 0, August 2024.

[6] Holtec Britain, "HI-2240337, Holtec SMR GDA PSR Part B Chapter 2 Reactor" Revision 0, August 2024.

[7] Holtec Britain, "HI-2240777, Holtec SMR GDA PSR Part B Chapter 5 Reactor Supporting Facilities," Revision 0, August 2024.

[8] Holtec Britain, "HI-2240336, Holtec SMR GDA PSR Part A Chapter 5 Summary of ALARP," Revision 0, August 2024.

[9] Holtec Britain, "HI-2240335, Holtec SMR GDA PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance," Revision 0, August 2024.

[10] Holtec Britain, "HI-2240340, Holtec SMR GDA PSR Part B Chapter 9 Conduct of Operations," Revision 0, August 2024.

[11] Holtec Britain, "HI-2240341, Holtec SMR GDA PSR Part B Chapter 10 Radiological Protection," Revision 0, August 2024.

[12] Holtec Britain, "HI-2240344, Holtec SMR GDA PSR Part B Chapter 13 Radioactive Waste Management," Revision 0, August 2024.

[13] Holtec Britain, "HI-2240345, Holtec SMR GDA PSR Part B Chapter 14 Design Basis Accident Analysis," Revision 0, August 2024.

[14] Holtec Britain, "HI-2240346, Holtec SMR GDA PSR Part B Chapter 15 BDBA, Severe Accidents Analysis and Emergency Preparedness," Revision 0, August 2024.

[15] Holtec Britain, "HI-2240347, Holtec SMR GDA PSR Part B Chapter 16 Probabilistic Safety Assessment," Revision 0, August 2024.

[16] Holtec Britain, "HI-2240348, Holtec SMR GDA PSR Part B Chapter 17 Human Factors," Revision 0, August 2024.

[17] Holtec Britain, "HI-2240356, Holtec SMR GDA PSR Part B Chapter 19 Mechanical Engineering," Revision 0, August 2024.

[18] Holtec Britain, "HI-2240350, Holtec SMR GDA PSR Part B Chapter 21 External Hazards," Revision 0, August 2024.

[19] Holtec Britain, "HI-2240351, Holtec SMR GDA PSR Part B Chapter 22 Internal Hazards," Revision 0, August 2024.

[20] Holtec Britain, "HI-2240352, Holtec SMR GDA PSR Part B Chapter 23 Reactor Chemistry," Revision 0, August 2024.

[21] Holtec Britain, "HI-2240354, Holtec SMR GDA PSR Part B Chapter 25 Construction and Commissioning," Revision 0, August 2024.

[22] Holtec Britain, "HI-2240355, Holtec SMR GDA PSR Part B Chapter 26 Decommissioning Approach," Revision 0, August 2024.

[23] ONR, "Safety Assessment Principles for Nuclear Facilities," Revision 1, 2014 Edition, January 2020.

[24] U.S. Nuclear Regulatory Commission, "RG 1.26, Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants," Revision 6, December 2021.

[25] U.S. Nuclear Regulatory Commission, "Regulatory Guides," [Online] Available at: https://www.nrc.gov/reading-rm/doc-collections/reg-guides/index.html#guides.

[26] Holtec Britain, "HI-2240124, UK GDA Gap Analysis Report," Revision 0, February 2024.

[27] Holtec Britain, "HI-2240127, US/UK Regulatory Framework and Principles Report," Revision 0, February 2024.

[28] Holtec International, "Higher Reliability Identification Methodology," To be Issued.

[29] “Holtec Britain,"SMR-300 Generic Security Report for the Generic Design Assessment," Revision 0, May 2024”.

[30] American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code (BPVC), Section III – Rules for Construction of Nuclear Facility Components, Division 1 and Division 2," 2017.

[31] American Society of Mechanical Engineers, "ASME B16.34, Valves-Flanged, Threaded, and Welding End," 2020.

[32] American Society of Mechanical Engineers, "ASME B31.1, Power Piping," 2022 Edition.

[33] American Society of Mechanical Engineers, "ASME NQA-1, Quality Assurance Requirements for Nuclear Facility Applications," 2015.

[34] American Society for Testing and Materials, "ASTM E185-82, Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," July 1982.

[35] U.S. Nuclear Regulatory Commission, "NRC Regulations Title 10, Code of Federal Regulations, Appendix A to Part 50—General Design Criteria for Nuclear Power Plants".

[36] U.S. Nuclear Regulatory Commission, "NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Plants: LWR Edition,".

[37] TEMA, "Standards of the Tubular Exchanger Manufacturers Association," 10th Edition, 2019.

[38] R. Bullough, F. M. Burderkin, O. V. J. Chapman, V. R. Green, D. P. G. Lidbury, J. N. Swingler and R. Wilson, "The demonstration of incredibility of failure in structural integrity safety cases," *International Journal of Pressure Vessels and Piping,* no. 78, pp. 539-552, August 2001.

[39] U.S. Nuclear Regulatory Commission, "NUREG-0800: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition - Chapter 3: Design of Structures, Components, Equipment, and Systems - Section 3.2.2: System Quality Group Classification," Revision 3, July 2016.

[40] American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code, Section III – Rules For Construction of Nuclear Facility Components," 2021.

[41] American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code, Section II - Materials," 2023 & 2021 .

[42] U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.44, Control of the Processing and use of Stainless Steel," March 2011, Revision 1.

[43] U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.28 Quality Assurance Program Criteria (Design and Construction)," October 2017.

[44] U.S. Nuclear Regulatory Guide, “Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation),” June 2013.

[45] Holtec International, "HI-2230318, Decision Paper on Corrosion Allowance in SMR-160 Piping," Revision 0, 2023.

[46] U.S. Nuclear Regulatory Commission, “NRC Regulations Title 10, Code of Federal Regulations, Appendix G to Part 50—Fracture Toughness Requirements”.

[47] EDF Energy, “R6 Assessment of the Integrity of Structures Containing Defects, Revision 4, Amendment 12,” 2019.

[48] Holtec International, "HI-2230815, Topical Report on The Quality Assurance Program for Holtec International's Small Modular Reactor (SMR) Design and Construction," Revision 0, September 2023.

[49] American Society of Mechanical Engineers, “Boiler and Pressure Vessel Code, Section IX – Qualification Standard for Welding,” 2023 & 2021.

[50] American Society of Mechanical Engineers, “Boiler and Pressure Vessel Code, Section V – Non-destructive Examination,” 2023 & 2021.

[51] American Society of Mechanical Engineers, “Boiler and Pressure Vessel Code, Section VIII - Rules for Construction of Pressure Vessels,” 2023.

[52] Holtec International, "CD-03, Nuclear Quality Assurance Manual," Revision 15, 2022.

[53] Holtec International, "HI-2125399, QA Validation of the R-CODE R6 Module," Revision 2, 2019.

[54] NUGENIA Association, “ENIQ Methodology Document: European Methodology for Qualification of Non-Destructive Testing: Report Number 61,” March 2019.

[55] American Society of Mechanical Engineers, “Boiler and Pressure Vessel Code, Section XI – Rules for Inservice Inspection of Nuclear Power Plant Components,” 2023 & 2021.

[56] U.S. Nuclear Regulatory Commission, “NRC Regulations Title 10, Code of Federal Regulations Appendix H to Part 50—Reactor Vessel Material Surveillance Program Requirements”.

[57] Holtec Britain, "HPP-3295-0017, Holtec SMR-300 Generic Design Assessment Reference Design Process and GDA Prospective Design Change Register," Revision 0, May 2024.

[58] U.S. Nuclear Regulatory Commission, “Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” March 2007, Revision 1.

18.12 LIST OF APPENDICES

Appendix A Structural Integrity CAE Route Map

Table 4: Part B Chapter 18 CAE Route Map

Overarching SSEC Claim	PSR Part B Chapter 18 Structural Integrity Claim	SI Chapter Sub-claims	Arguments
Claim 2.2 – System / Process Design and Substantiation: **The design of the systems and associated processes have been developed taking cognisance of relevant good practice and substantiated to achieve their safety and non-safety functional requirements.**	**Claim 2.2.9** – Higher Reliability SSCs have been justified using appropriate methods, demonstrating that risk is tolerable and As Low As Reasonably Practicable (ALARP).	2.2.9.1 - Structural Integrity SSCs are designed using appropriate Codes and Standards.	SSCs will be designed utilising relevant codes, RGP, and OPEX.
		2.2.9.2 - Higher reliability SSCs Structural Integrity is achieved through quality design, analysis, manufacture, and testing.	Appropriate materials will be selected with consideration given to compatibility and representative testing.
			Analysis of Structural Integrity will be conducted utilising conservative parameters.
			Where the principal of Defence in Depth cannot be employed, 'Conceptual' Defence in Depth will be evoked.
		2.2.9.3 - Higher reliability components will be tolerant of defects demonstrated by the avoidance of fracture.	Defect tolerance assessment utilising conservative material properties, fracture toughness testing, and through life inspections will be utilised to demonstrate the avoidance of fracture.
			Manufacturing will be aligned with relevant codes and will build on lessons learnt from OPEX and RGP.
		2.2.9.4 - Through life monitoring, maintenance and inspection will provide forewarning of failures.	Design will consider future plant operation, maintenance, and ability to be inspected.